10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

May 29, 2008

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 07-0843A
NAPS: MPW
Docket No.: 50-339
License No.: NPF-7

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following revised Licensee Event Report applicable to North Anna Power Station Unit 2.

Report No. 50-339/2007-004-01

This report has been reviewed by the Facility Safety Review Committee and will be forwarded to the Management Safety Review Committee for its review.

Sincerely,

Dan Stoddard

Daniel G. Stoddard, P.E.
Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Senior Resident Inspector
North Anna Power Station

IE22
NRR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Automatic Reactor Trip Due to Loss of Coolant Flow With Power Greater Than 30 Percent

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000
12	25	2007	2007	-- 004 --	01	05	29	2008	FACILITY NAME	DOCUMENT NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (*Check all that apply*)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	VOLUNTARY LER

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (*Include Area Code*)
F. Mladen, Director Station Safety and Licensing	(540) 894-2108

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	RC	P	W120	Y	A	BA	P	I238	Y

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*) ☒ NO				

ABSTRACT (*Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines*)

On December 25, 2007 at 2110 hours, with Unit 2 in Mode 1 at 100% power, an automatic reactor trip occurred. The automatic reactor trip was a result of a loss of coolant flow with power greater than 30 percent. The loss of coolant flow was the result of the B Reactor Coolant Pump (RCP) motor trip. A review of the event determined that the breaker supplying the B RCP motor opened due to a response by the neutral over current protection relay. Post-event testing of the B RCP motor trip identified a "B" phase to ground fault. The ground fault of the motor winding was caused by a manufacturing defect leading to premature aging of the coil insulation system. This event is reportable pursuant to 10 CFR 50.73(a)(2)(iv)(A) for an event that resulted in the automatic actuation of the Reactor Protection System and Engineered Safety Features Actuation Systems. All Engineered Safety Feature equipment responded as designed with the exception of the steam driven auxiliary feedwater pump which tripped due to actuation of the over-speed trip valve and was manually reset and subsequently placed in service. The health and safety of the public were not affected at any time during the event.

1.0 DESCRIPTION OF THE EVENT

At 2110 hours on 12/25/2007, Unit 2 experienced an automatic reactor trip due to loss of reactor coolant (RCS) flow with power greater than 30 percent. A review of the event determined that the motor breaker supplying the Unit 2 B Reactor Coolant Pump (RCP) motor (EIIS System-AB, Component-MO) opened due to a response by the neutral over current protection relay (EIIS Component-RLY).

The digital fault recorder displayed a peak neutral current of 2189 amps for approximately 100 milliseconds. The fault was on "B" phase to ground. This was cleared by the motor feeder breaker opening. A post-event insulation to ground test of the field cables between the motor breaker (EIIS Component-BKR) and the B RCP motor determined that no damage had been incurred and that the field cables remained acceptable for service.

At 2322 hours on 12/25/2007, a 4-hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B), for actuation of the Reactor Protection System (RPS). An 8-hour Non-Emergency Report was also made in accordance with 10 CFR 50.72 (b)(3)(iv)(A) for an actuation of the Auxiliary Feedwater System.

Unit 2 was placed in Mode 5, cold shutdown, to facilitate removal and replacement of the B RCP motor. On site evaluation of the B RCP motor determined that the fault occurred to the motor stator assembly. Subsequent inspection by the vendor identified damage to the rotor. The subject motor was removed from the Unit 2 containment building and replaced with a spare motor. The faulted motor assembly was shipped to a vendor where a motor failure analysis was performed. North Anna personnel were directly involved in the motor failure analysis.

The B RCP motor is a Westinghouse Model CS, S/N 1S81P777, 7000 HP, 4000 volt, 1185 RPM, Full load amp 891, 6 pole, Class H windings and a service factor of 1.15. The motor windings were replaced in 1992. A vendor inspection/refurbishment was last performed in 1999. At that time, corona activity was identified on all of the line coils of the stator winding. The vendor treated the windings to preclude reoccurrence. Since refurbishment in 1999, the motor has been in service as the 2-RC-P-1B motor. A review of motor stator temperature and vibration readings prior to the event did not reveal any abnormalities. The review of all Preventive Maintenance (PM) test and inspection data from the past two refueling outages for B RCP motor determined that there have been no electrical or mechanical abnormalities. The electrical readings taken in the spring of 2007 for the motor stator windings compared closely to the ones taken in fall of 2005 and showed no evidence of degradation.

During the event the steam driven auxiliary feedwater pump tripped due to actuation of the over speed trip valve. The steam driven auxiliary feedwater pump (EIIS System- BA, Component-P) was manually reset and placed in service.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

There is no impact on the design function, ability to function and method of performing the function or control/operation of a structure, system or component (SSC) described in the USFAR. Reactor coolant pumps are designed to provide core cooling and not required during emergency operating conditions. The ability to maintain the reactor shutdown is not affected by failure of these components.

Following the reactor trip the Reactor Protection System (RPS) and all Engineered Safety Feature Actuation System (ESFAS) (EIIS System JE) equipment responded as designed with the exception of the Auxiliary Feedwater System (AFW) (EIIS System BA) steam driven pump. The steam driven auxiliary feedwater pump was reset manually and operated as designed. No other major equipment issues were noted. This event is reportable pursuant to 10 CFR 50.73(a)(2)(iv)(A) for an event that resulted in the automatic actuation of the Reactor Protection System and Engineered Safety Features Actuation Systems. The health and safety of the public were not affected at any time during the event.

3.0 CAUSE

The reactor trip was the result of a loss of coolant flow with power greater than 30 percent. The B RCP motor trip was a result of a “B” phase to ground fault.

A root cause evaluation (RCE) has verified the B RCP motor trip was the result of a “B” phase to ground fault. The ground fault of the motor winding was caused by a manufacturing defect leading to premature aging of the coil insulation system. The root cause was attributed to a defective Vacuum Pressure Injection (VPI) process, performed during the 1992 rewind of the motor by the vendor.

Failure to fully submerge the stator winding in the resin prevented the coils from being tightly secured in its core slot. This will allow the coils to vibrate under magnetically induced mechanical forces resulting in movement between the winding ground insulation outer surface and the core. Being somewhat serrated, the coil abrades the semi-conductive coating and then the ground wall insulation. If not detected, this insulation degradation can result in a winding ground fault.

The extent of the cause could include all motors which have had the VPI process to the windings. The RCE investigation has determined that this failure is limited to a specific vendor's process. Therefore, there is no evidence to suggest that this issue extends beyond the RCP motors. Preventative and Predictive analysis is the means by which trending for this type of deficiency can be obtained for all motors.

The steam driven auxiliary feedwater pump trip was caused by the method used for performing certain critical dimensional checks that should be performed when assembling

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

the trip linkage for the overspeed trip valve. The measurement method used had been an accepted standard with the vendor technical representatives that were used at North Anna. Additional information obtained from a plant with a similar operating experience led to a more accurate method for obtaining critical dimension checks.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the event in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Control Room personnel stabilized the plant using ES-0.1 Reactor Trip Recovery. All safety systems responded appropriately with the exception of the steam driven auxiliary feedwater pump which tripped due to actuation of the over speed trip valve and was manually reset and subsequently placed in service. The unit was stabilized at no-load conditions.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A failure analysis for the B RCP motor was completed. The motor will be refurbished. The steam driven auxiliary feedwater pump trip linkage was rebuilt to restore manufacturer recommended tolerances. Training has been provided to the Operations staff on the enhancements for latching the trip valve linkage and emergency trip mechanism engagement.

6.0 ACTIONS TO PREVENT RECURRENCE

The specification for motor refurbishments is being revised to include proper verification and testing of completed Vacuum Pressure Injection (VPI) treatment. The specification will also include recommendations related to the VPI process per EPRI TR 1009700, Guide for Electric Motor Stator Winding Insulation Design, Testing and VPI Resin Treatment.

Maintenance procedures for the steam driven auxiliary feedwater pump have been revised to provide additional detail to ensure the emergency trip mechanism engagement meets the manufacturer recommended tolerances.

7.0 SIMILAR EVENTS

There have been no similar events where the reactor tripped on a loss of coolant flow with power greater than 30 percent as a result of an RCP motor trip.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

8.0 ADDITIONAL INFORMATION

At the time of this event, North Anna Unit 1 was operating at 100 percent power and was not affected by this event.

Component information:
Description: Reactor Coolant Pump Motor
Manufacturer: Westinghouse
Model No.: CS
Serial No.: 1S81P777